

ACCESSION #: 9505020150  
LICENSEE EVENT REPORT (LER)

FACILITY NAME: PILGRIM NUCLEAR POWER STATION PAGE: 1 OF 9

DOCKET NUMBER: 05000293

TITLE: Manual Scram Due to Main Generator Stator Cooling Water  
Temperature Control Valve Controller Failure  
EVENT DATE: 03/24/95 LER #: 95-003-00 REPORT DATE: 04/24/95

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 60

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR  
SECTION:  
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:  
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Compliance Engineer

COMPONENT FAILURE DESCRIPTION:  
CAUSE: X SYSTEM: TJ COMPONENT: TC MANUFACTURER: F120  
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

#### ABSTRACT:

On March 24, 1995 at 0750 hours, a manually initiated Reactor Protection System (RPS) scram signal and scram occurred while at 60 percent reactor power. The signal resulted in an automatic sequence of expected designed responses that included a Turbine-Generator trip and an automatic transfer of station electrical loads.

The direct cause for the scram signal was the deliberate movement of the reactor mode selector switch from the RUN position while at 60 percent reactor power. This action was taken due to increasing Main Generator stator cooling water (SCW) temperature and conductivity, increasing generator stator temperatures, and a generator field ground alarm. The root cause for the event was a failure of the mechanical linkage of the SCW temperature control valve controller. Corrective action planned includes replacement of the temperature control valve controller and linkage.

This event occurred during power operation. The Reactor Vessel (RV) pressure was 982 psig with the RV water temperature at approximately 544 degrees Fahrenheit. This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) and the event posed no threat to the public health and safety.

END OF ABSTRACT

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## BACKGROUND

The Turbine-Generator is non-safety related and consists of the turbine, generator, exciter, controls, and required subsystems. The turbine utilizes mechanical hydraulic controls and converts steam to rotational energy. The principal cooling medium is hydrogen gas that is contained within the generator frame and circulated by fans mounted at each end of the generator rotor. The stationary armature winding (stator) is cooled externally by hydrogen gas and internally by cooling water. The stator bars are composed of hollow insulated copper conductors, arranged in the form of rectangular bars. An insulation system is applied over each stator bar. During generator operation, cooling water flow through the stator bars prevents overheating of the insulation and stator bars that could lead to generator damage.

The cooling water flow through the stator bars is clean, low conductivity water provided by the Stator Water Cooling (SCW) subsystem. The SCW subsystem is designed to operate as a subloop of the Turbine-Generator control system. Provision is made for automatic regulation of SCW temperature and flow to the stator windings and the generator excitation system rectifiers.

The SCW subsystem includes instrumentation, a storage tank, two pumps connected in parallel, two in-series coolers, filters, a demineralizer, valves, and piping to and from the generator stator windings and excitation system rectifiers. During normal operation, SCW flow to the stator windings and exciter rectifiers is drawn from the storage tank, pumped through the coolers, filtered, directed to the stator windings and rectifiers, and returned to the storage tank. A portion of the SCW flow is directed to the demineralizer for continuous purification of the SCW water. Some of the SCW flow is directed around the coolers by temperature control valve TCV-Y07 that functions to control SCW temperatures. This bypass flow is mixed with the cooler water from the SCW coolers at the cooler outlet. If the SCW temperature increases to approximately 87 degrees Celsius, an automatic Turbine runback signal

would be generated and cause the Turbine-Generator control system to decrease generator electrical load to prevent Turbine-Generator damage. If the load does not decrease to a specific setpoint within approximately 3.5 minutes or SCW temperature does not decrease below 87 degrees Celsius, an automatic Turbine trip signal (MTS-1) is generated.

Stator bar temperatures are monitored by the Generator Monitoring System that was installed during the 1994 generator outage. This system provides alarm and indication in the Main Control Room.

On March 24, 1995, at 0738 hours, a Main Generator monitoring trouble alarm (Panel C-3R window C1) occurred. Operator response to the alarm included checking parameters, including generator stator and SCW temperatures. The monitoring system indicated increasing SCW temperatures. Reactor power was manually decreased to approximately 90% via recirculation pump speed control.

At 0741 hours, a Main Generator cooling water trouble alarm (Panel C3R window D1) occurred. Operator response to the alarm included checking Panel C100 status. The operator reported a high SCW cooling water temperature alarm with SCW temperature indicating 80 degrees Celsius at Panel C100.

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At 0746 hours, a Main Generator field ground alarm (Panel C3R window F1) occurred. Reactor power was manually decreased to 80% via recirculation pump speed control. Operators at Panel C100 and the SCW cooler reported SCW temperature rising at 90 degrees Celsius, and reported temperature control valve TCV-Y07 was not operating correctly. The operator attempted to manually control TCV-Y07. SCW conductivity was also noted to be increasing.

Reactor power was further decreased to approximately 60% by decreasing the recirculation pump speeds. The Nuclear Watch Engineer (NWE) briefed the crew on a manual scram and ordered the scram. Coincident with the order, a Main Generator runback alarm (C2L window E2) was received.

Just prior to the scram, plant operating conditions included the following:

- o The reactor mode switch was in the RUN position. The reactor power was 60 percent. The Reactor Vessel (RV) pressure was 982 psig with the RV water temperature at approximately 544 degrees Fahrenheit. The RV water level was approximately +28 inches.

- o The Recirculation System motor-generator sets/pumps 'A' and 'B' were in service with each loop in the local manual control mode. Core flow was 33 Mlb/hr after being reduced from 72 Mlb/hr by the Control Room Operator. The Condensate System and Feedwater System pumps were all in service. The Feedwater Level Control System was in the three element control mode.
- o The 345 KV transmission lines 342 and 355 were energized. ACBs-102, -103, -104, and -105 were closed. The 4160 VAC Auxiliary Power Distribution System (APDS) was energized from the Unit Auxiliary Transformer (UAT) with the bus transfer switches in the ON position. The Startup Transformer (SUT) was in standby service. The Shutdown Transformer was in standby service with the 23 KV distribution system energized.
- o The Emergency Diesel Generators 'A' and 'B' were in standby service.

#### EVENT DESCRIPTION

On March 24, 1995 at 0750 hours, a manually initiated Reactor Protection System (RPS) scram signal and scram occurred while at 60 percent reactor power. The scram was the result of the deliberate movement of the reactor mode switch from the RUN position to the SHUTDOWN position. This action was taken due to increasing SCW temperature, increasing SCW conductivity, increasing Main Generator stator temperatures, and a Main Generator field ground alarm.

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As expected, the scram signal resulted in an automatic sequence of designed responses that included a Turbine-Generator trip. The Turbine trip included the following responses:

- o Automatic closing of the Main Steam System/Turbine Valves (stop valves, control valves, combined intermediate valves), automatic opening of the Turbine Bypass valves, and a trip of the Turbine lockout relay (286-2).
- o Automatic opening of the Generator Field Breaker. The Generator trip was the designed response to the loss of field that resulted from the automatic opening of the field breaker.
- o Automatic opening of the 345 kV switchyard circuit breakers ACB-104 (352-4) and ACB-105 (352-5).
- o Automatic transfer of the source of power for APDS from the UAT to

the SUT.

The RV water level decreased to approximately -38 inches in response to the decrease in the void fraction in the RV water. The RV water level decrease to below +12 inches resulted in the automatic actuation of the Primary Containment Isolation Control System (PCIS) and Reactor Building Isolation Control System (RBIS).

The PCIS actuation resulted in the following designed responses:

- o Automatic closing of the inboard and outboard Primary Containment System (PCS)/Reactor Water Sample isolation valves AO-220-44 and -45.
- o Automatic closing of the inboard and outboard PCS Group 2 isolation valves that were open.
- o The PCIS Group 3/Residual Heat Removal (RHR) System Shutdown Cooling suction piping isolation valves MO-1001-47 and -50 remained closed.
- o The PCS Group 3/RHR System Low Pressure Coolant Injection mode valves MO-1001-29A and -29B remained closed.
- o The PCS Group 6/Reactor Water Cleanup (RWCU) System isolation valves closed automatically.

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The RBIS actuation resulted in the automatic start of the Standby Gas Treatment System (SGTS) Trains 'A' and 'B' and automatic closing of the Reactor Building/Secondary Containment System (SCS) Trains 'A' and 'B' supply and exhaust ventilation dampers except for supply damper AON-81. The position indication for the in-series supply dampers AON-80 and AON-81 both indicated an intermediate position (i.e., OPEN and CLOSED).

The initial Control Room operator response was orderly. The reactor mode switch was moved to the REFUEL position in accordance with procedure 2.1.6, "Reactor Scram". EOP-01, "RPV Control", was entered at 0752 hours when the RV water level lowered below +12 inches. The verification of the insertion of all control rods was initiated and completed promptly after the scram.

The Control Room Panel C7 control switches for dampers AON-80 and AON-81 were manipulated to close the dampers in accordance with EOP-01. Local indication of secondary containment pressure was -0.3 inches of water with SGTS in service. The indicated positions of AON-80 and AON-81

continued to indicate an intermediate position. A local inspection of damper AON-80 determined the damper was fully closed and a closed position indication was obtained after the limit switch was tapped. Damper AON-81 was found approximately 75 percent closed and was manually shut.

An indication of a partial PCIS Group 1 isolation signal was observed. This observation was made by a Licensed Operator in the Control Room in that a logic channel light was not lit.

The RPS was reset at 0801 hours.

At 0811 hours, the RWCU/Group 6 isolation was reset and the system returned to service.

EOP-04, "Secondary Containment Control", was entered at 0814 hours due to the Main Steam Tunnel temperature greater than 120 degrees Fahrenheit and HPCI Piping Area temperatures greater than 105 degrees Fahrenheit. Radiation Protection personnel were notified to initiate a radiation survey in the Reactor Building in accordance with EOP-04. Survey results were normal, and EOP-04 was exited at 0935 hours after the temperatures decreased to below the maximum normal operating values of EOP-04.

At 0821 hours, an operator noted some SCW leakage from the Main Generator rectifier bank. The leak was isolated locally.

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The RV pressure was maintained at approximately 940 psig pending Drywell de-inerting and subsequent licensed operator entry into the Drywell for planned inspections.

The primary containment de-inerting was commenced at 0905 hours.

At 0917 hours, EOP-01 was exited.

Portions of the Group 2 PCIS were reset at 1036 hours.

An unplanned actuation of the Drywell-to-Torus vacuum breakers occurred at 1101 hours while de-inerting the Drywell. The event is separately reported via LER 95-004.

The SGTS fans 'A' and 'B' subsequently tripped at different times during de-inerting. The SGTS 'A' fan tripped at 1423 hours and the SGTS 'B' fan was then placed in service. After removing the 'B' fan from the Drywell purge lineup, for the performance of Procedure 8.7.2.6, "SGTS Single

Train Operability Test", the 'B' fan would not start at 1540 hours. The SGTS Train 'A' was then surveillance tested with satisfactory results. SGTS train 'B' was subsequently tested satisfactorily via Procedure 8.7.2.6. No other anomalies were observed with SGTS fans. Procedure 8.7.2.6 verifies fan operability by simulating the SGTS train safety function lineup. Problem Reports (PR)95.9141 and PR95.9189 were written for the SGTS fan trips.

The reactor mode switch was moved from the REFUEL position to the SHUTDOWN position at 2111 hours. This action resulted in an expected RPS scram signal. The control rods remained fully inserted.

At 2113 hours, the RPS was reset.

Procedure 2.1.7 (Rev. 30), "Vessel Heatup and Cooldown", was initiated at 0530 hours on March 25, 1995.

Vessel cooldown was commenced at 1104 hours on March 25, 1995. After preparations that included flushing of applicable piping, the Residual Heat Removal (RHR) System Loop 'B' was placed in shutdown cooling mode (SDC) at 1821 hours. The RV temperature was less than 212 degrees Fahrenheit by 1900 hours.

At 1940 hours, the RV head vent valves were opened.

PR95.9150 was written to document the failure of TCV-Y07. PR 95.9140 was written to document the problem with the position indication for AON-80 and incomplete closing of damper AON-81. The NRC Operations Center was notified in accordance with 10 CFR 50.72 at 0950 hours on March 24, 1995. Several other Problem Reports were written to document other aspects or observations related to the event.

A post trip review of the event was initiated in accordance with Procedure 1.3.37 (Rev. 10), "Post Trip Reviews".

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A critique was held on March 24, 1995. The critique was attended by appropriate personnel including licensed operators on-shift at the time of the event.

## CAUSE

The direct cause of the scram was the intentional movement of the reactor mode switch from the RUN position while at 60 percent reactor power. A high neutron flux RPS trip signal was generated as expected when the

reactor mode switch passed through the STARTUP position due to reactor power being greater than the 15 percent high flux trip setpoint. Another RPS trip signal was generated as expected when the reactor mode switch was placed in the SHUTDOWN position.

The controller for TCV-Y07 was manufactured by the Fischer & Porter Company, model number 1450. The controller was installed during the 1993 refueling outage.

The root cause of the event was a premature failure of the mechanical linkage of the non-safety related TCV-Y07 controller. The linkage consisted of a metal rod threaded into a nylon connector. The rod became disengaged from the nylon connector. This caused TCV-Y07 to open fully and resulted in the maximum amount of bypass flow around the SCW coolers, increased SCW temperature and conductivity, increased generator stator temperature, and alarms. The cause of damper AON-81 not fully closing was the fouling of the actuator by dirt.

The cause of the SGTs fans tripping during drywell purge operations was the SGTs heaters' high temperature trip setpoint was set at 150 degrees Fahrenheit. An evaluation is currently being performed to possibly increase the setpoint. The ability of the SGTs fans to perform their secondary containment function was demonstrated at the time by successful performance of Procedure 8.7.2.6.

The observation regarding the partial PCIS Group 1 isolation signal is being evaluated (PR95.9189). The PCIS logic circuitry is normally energized and is designed to de-energize for the isolation function. The observed condition would not have prevented the isolation function from occurring.

## CORRECTIVE ACTION

The controller of TCV-Y07, including the linkage, is no longer manufactured and is no longer available. The controller and linkage are being replaced.

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Corrective actions related to the Secondary Containment Dampers, including damper AON-81, have been identified in PR95.9140. The dampers were added to a preventive maintenance schedule. The Preventive Maintenance plan for the SCS dampers is to test the dampers, and rebuild the dampers' actuators if necessary, every 4-5 years of service. The actuator of damper AON-81 was cleaned, rebuilt, and bench tested satisfactorily during the 1995 refueling outage. All other SCS dampers'

actuators are currently scheduled to be rebuilt during the 1995 refueling outage.

## SAFETY CONSEQUENCES

This event posed no threat to public health and safety.

The design basis of the SCS, to be sufficiently leak tight to allow the SGTS to reduce the Reactor Building sub-atmospheric pressure to at least -0.25 inches of water, was met. The Licensed Operator verified by direct observation the Reactor Building sub-atmospheric pressure was -0.30 inches of water with the SGTS in service and the SCS dampers including AON-80 and AON-81 in the as-found condition, i.e. damper AON-80 fully closed and in-series damper AON-81 approximately 75 percent closed.

The decrease in the RV water level was the expected response to the scram and accompanying shrink in the RV water. The PCIS and RBIS actuations were the expected responses to a low RV water level condition (i.e., less than approximately +12 inches).

The setpoint for automatic actuation of the Core Standby Cooling Systems (CSCS) is approximately -46 inches. During the event, the lowest RV water level that occurred (-38 inches) was approximately 8 inches above the CSCS setpoint. In addition, the level was approximately 89.5 inches above the level that corresponds to the top of the active fuel zone. The lowest RV water level that occurred was also greater than the setpoint (approximately -46 inches) that initiates the ATWS System functions for a Recirculation Pump Trip (RPT) and Alternate Rod Insertion (ARI).

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) because the actuation of the RPS, although a designed response to the intentional movement of the reactor mode switch from the RUN position while at 60 percent reactor power, was not planned.

## SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports (LERs) submitted since January 1984. The review focused on LERs submitted in accordance with 10 CFR 50.73(a)(2)(iv) involving a similar manual scram or problem with the SCW system. The review identified LER 94-005-00.

For LER 94-005-00, an automatic scram occurred on August 29, 1994, at 0732 hours while at 100 percent reactor power. The scram was the result of a load rejection that was automatically initiated by a Main Generator stator faulted condition. Corrective action taken included the replacement of the Main Generator stator.

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## ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

### COMPONENTS CODES

Damper (AON-81) DMP

Valve, Control, Temperature (TCV-Y07) TCV

### SYSTEMS

Stator Cooling Water System TJ

Engineered Safety Features Actuation System JE  
(PCIS, RBIS, RPS)

Standby Gas Treatment System BH

Containment Vacuum Relief System BF

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### BOSTON EDISON

Pilgrim Nuclear Power Station 10 CFR 50.73

Rocky Hill Road

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E. T. Boulette, PhD

Senior Vice President -Nuclear

April 24 1995

BECo Ltr. #95-053

U.S. Nuclear Regulatory Commission

Attn: Document Control Desk

Washington, D.C. 20555

Docket No. 50-293

License No. DPR-35

The enclosed Licensee Event Report (LER) 95-003-00, "Manual Scram Due to Main Generator Stator Cooling Water Temperature Control Valve Controller Failure", is submitted in accordance with 10 CFR Part 50.73.

Please do not hesitate to contact me if there are any questions regarding this report.

E. T. Boulette, PhD

TRD/lam/9500300

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